ACCESSION #: 9701140322

LICENSEE EVENT REPORT (LER)

FACILITY NAME: SURRY POWER STATION, Unit 2 PAGE: 1 OF 4

DOCKET NUMBER: 05000281

TITLE: Auto Rx Trip due to Stm/Feed Flow Mismatch Coincident

with a Low S/G Level

EVENT DATE: 12/13/96 LER #: 96-006-00 REPORT DATE: 01/02/97

OTHER FACILITIES INVOLVED: DOCKET NO: 05000

OPERATING MODE: N POWER LEVEL: 11%

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION:

50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:

NAME: D. A. Christian, Station Manager TELEPHONE: (757) 365-2000

COMPONENT FAILURE DESCRIPTION:

CAUSE: SYSTEM: COMPONENT: MANUFACTURER:

REPORTABLE NPRDS:

SUPPLEMENTAL REPORT EXPECTED: NO

ABSTRACT:

At 0233 hours on December 13, 1996, with Unit 2 at 11% power, an automatic reactor trip occurred during a planned shutdown. The reactor trip was caused by a steam flow/feedwater flow mismatch in coincidence with low Steam Generator (S/G) level in the 'A' S/G.

Upon receipt of the automatic reactor trip, the Reactor Protection System (RPS) functioned as designed and rod bottom lights were lit for all control rods except 2-RD-RPI-P-6. All electrical buses remained energized by off-site power and all emergency diesel generators were operable. Station operating personnel promptly placed the plant in

a stable, hot shutdown condition in accordance with the. appropriate procedures. The shutdown margin was calculated and found to be satisfactory. The health and safety of the public were not affected by this event since all plant parameters remained within the normal range and all required safety equipment operated as designed.

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1.0 DESCRIPTION OF THE EVENT

At 0233 hours on December 13, 1996, with Unit 2 at 11% power, an automatic reactor trip occurred during a planned shutdown. The reactor trip was caused by a steam flow/feedwater flow mismatch in coincidence with low Steam Generator (S/G) level in the 'A' S/G [EIIS-JE]. Automatic actuation of the Reactor Protection System (RPS) and Engineered Safeguards Features (ESF) occurred as expected, including Turbine Trip by Reactor Trip and Auxiliary Feedwater initiation. Station operating personnel promptly placed the plant in a stable, hot shutdown condition in accordance with the appropriate procedures. The shutdown margin was calculated and found to be satisfactory. The Shift Technical Advisor monitored the critical safety function status trees to verify that satisfactory unit conditions were maintained.

In accordance with 10CFR50.72(b)(2)(ii), a 4-hour Non-Emergency Report to the NRC operations center was made at 0425 hours due to the Reactor Protection System (RPS) and automatic ESF actuations. This event is being reported pursuant to 10 CFR 50.73(a)(2)(iv).

Following completion of the repairs during the planned shutdown, Unit 2 was taker, critical at 0348 hours and was placed on the line at 1243 hours on December 23, 1996. The unit was returned to 1 00% reactor power, 858 MWe at 1548 hours on December 24, 1996.

2.0 SIGNIFICANT SAFETY CONSEQUENCES AND IMPLICATIONS

Upon receipt of the automatic reactor trip, the RPS functioned as designed and rod bottom lights were lit for all control rods except 2-RD-RPI-P-6. Additionally, four Individual Rod Position Indicators (IRPI) [EIIS-JD-ZI] (2-RD-RPI-D4, F-6, M-4 and P-6) indicated between 10 and 32 steps. Subsequent rod drop tests verified that all Control Rods fully inserted into the core and that the observed irregularities were limited to position indicators. The health and safety of the public were not affected by this event since all plant parameters remained within the normal range and all required safety equipment operated as designed.

3.0 CAUSE OF THE EVENT

The preliminary root cause of the reactor trip is determined to be the interface between the operator and the system used to control S/G water level [EIIS-JB] at low power. At low power levels, indication of feedwater flow to the steam generators is not available and feedwater control and steam dumps are in manual in accordance with operating procedures. While operations personnel are trained for this evolution and normally perform well, the operating team was unsuccessful in mitigating a level decrease prior to a reactor trip on steam flow greater than feedwater flow coincident with low S/G water level. Manual control of S/G level at low power is difficult due to feedwater flow and steam flow instrumentation limitations, and the dynamic steam generator operating characteristics. Although this is a

difficult evolution, it is routinely performed successfully. The last occurrence of a similar event not caused by equipment problems occurred in 1989.

The operator's response to the dynamics of the S/G prior to the trip are consistent with operating procedures, training and philosophy for this evolution. The preliminary root cause did not identify any inappropriate operator actions and no mechanical equipment failure contributed to the trip.

4.0 IMMEDIATE CORRECTIVE ACTION(S)

Following the automatic reactor trip, the operating tea m performed the immediate actions of Emergency Operating Procedure 2-E-0, "Reactor Trip or Safety Injection." The reactor was verified to be tripped (rod bottom lights lit except for P-6, reactor trip and bypass breakers open, neutron flux decreasing). However, IRPIs D-4, F-6, M-4 and P-6 did not indicate less than 10 steps initially. The four affected IRPIs indicated 32, 20, 12 and 18 steps, respectively. The Source Range NIs automatically re-energized as expected. The turbine was manually tripped at 0233 hours. The minimum FICS temperature of 539 degrees Fahrenheit was reached at 0240 hours. At 0257 hours, the team transitioned to 2-ES-0.1, "Reactor Trip Response." The Shift Technical Advisor monitored the Critical Safety Function Status Trees. The operating team paused at Step 4 of 2-ES-0.1 "Verify all control rods - less than or equal to 10 steps" and evaluated IRPI indications. Four IRPIs were identified to be between 10 and 32 steps. As directed by procedure, additional boron was injected into the RCS to compensate for the potential of the four control rods not being fully inserted into the core.

5.0 ADDITIONAL CORRECTIVE ACTION(S)

Hot rod drop tests (2-NPT-RX-014) were performed for all control rods. The data collected was satisfactory with no anomalies noted. Specifically, rod drop times were within the Technical Specification limits and were consistent with previous data. These test results confirmed that the control rods were fully inserted into the core and the observed irregularities were limited to IRPI problems. The following actions were taken to correct the IRPI problems:

2-RD-RPI-D4 is located in Control Bank C. Following the trip, this IRPI indicated 32 steps and its rod bottom light was illuminated. The rod bottom light is expected to illuminate at 20 steps. The rod bottom bistable was calibrated and found to be within specifications indicating that the actual rod position was below 20 steps. The IRPI pointer was found rubbing on the faceplate. The IRPI was replaced. Additionally, the signal conditioning module required a zero and span adjustment. The IRPI was calibrated satisfactorily and returned to service.

2-RD-RPI-F6 is located in Control Bank D. Following the trip, this IRPI indicated 20 steps. The signal conditioning module was replaced. The IRPI was calibrated satisfactorily and returned to service.

2-RD-RPI-M4 is located in Control Bank C. Following the trip, this IRPI indicated 12 steps. The signal conditioning module was calibrated satisfactorily to the baseline calibration voltages. The indication returned to 18 steps when the IRPI was returned to service. This behavior indicates that new calibration voltages were needed. At Hot Shutdown, new baseline calibration voltages were developed in accordance with the Analog Rod Position Indication System Channel Calibration procedure. The IRPI was calibrated satisfactorily using the new voltages and returned to service.

2-RD-RPI-P6 is located in Control Bank A. Following the trip, this IRPI indicated 18 steps and the rod bottom light was not illuminated. The signal conditioning module required zero and span adjustments. When the mechanical zero could not be adjusted, the IRPI was replaced. The rod bottom bistable was also found to trip out of tolerance, so the bistable was replaced. The IRPI was calibrated satisfactorily and returned to service.

6.0 ACTIONS TO PREVENT RECURRENCE

A Category I Root Cause Evaluation is being conducted to identify the direct and contributing causes of this event. Recommendations from the Root Cause Evaluation will be reviewed by management. Approved recommendations will be implemented.

7.0 SIMILAR EVENTS

LER S2-96-004-00, Unit 2 Trip Due to an Automatic High-High S/G Level Trip on the "B" S/G. This event was due to equipment problems with the Feedwater Regulating System.

8.0 ADDITIONAL INFORMATION

Unit 1 was operating at 100% and was not affected by this event.

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10CFR50.73

Virginia Electric and Power Company

Surry Power Station

5570 Hog Island Road

Surry, Virginia 23883

January 3, 1997

U. S. Nuclear Regulatory Commission Serial No.: 97-002

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Washington, D.C. 20555 Docket No.: 50-281

License No.: DPR-37

Dear Sirs:

Pursuant to Surry Power Station Technical Specifications, Virginia

Electric and Power Company hereby submits the following Licensee Event

Report applicable to Surry Power Station Unit 2.

REPORT NUMBER

50-281/96-006-00

This report has been reviewed by the Station Nuclear Safety and Operating

Committee and will be forwarded to the Management Safety Review Committee

for its review.

Very truly yours,

D. A. Christian

Station Manager

Enclosure

cc: Regional Administrator

101 Marietta Street NW Suite 2900

Atlanta, Georgia 30323

R. A. Musser

NRC Senior Resident Inspector

Surry Power Station

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